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Mr. Lew W. Myers
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

SUBJECT: AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY NUCLEAR POWER PLANT, UNIT 1 (TAC NO. MA2290)

Dear Mr. Myers:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 101 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit 1. This amendment revises the Technical Specifications in response to your application dated July 13, 1998 (PY-CEI/NRR-2298L), as supplemented by submittal dated November 23, 1998 (PY-CEI/NRR-2332L).

This amendment revises Technical Specification 3.4.4, "Safety/Relief Valves (SRVs)," by increasing the present ±1% tolerance on the safety mode lift setpoint for the safety relief valves to ±3%.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original signed by:
Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 101 to License No. NPF-58
2. Safety Evaluation

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*See G. Thomas to D. Pickett memo dated 01/15/99

** See C Berlinger to S Richards memo dated 02/01/99

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 3, 1998

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Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
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cc w/encls: See next page

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FirstEnergy Nuclear Operating Company

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. NPF-58

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee, formerly The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, and Toledo Edison Company) dated July 13, 1998, as supplemented by submittal dated November 23, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 101 are hereby incorporated into this license. The FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 90 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas V. Pickett

Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 3, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 101

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.4-10	3.4-10

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (S/RVs)

LCO 3.4.4 The safety function of seven S/RVs shall be OPERABLE.

AND

The relief function of six additional S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify the safety function lift setpoints of the required S/RVs are as follows: Number of S/RVs — Setpoint (psig) 8 — 1165 ± 34.9 6 — 1180 ± 35.4 5 — 1190 ± 35.7	In accordance with the Inservice Testing Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated July 13, 1998 (Ref.1), the FirstEnergy Nuclear Operating Company, the licensee for Perry, submitted proposed changes to Technical Specification (TS) section 3.4.4, "Safety/Relief Valves (SRVs)." The licensee submitted additional information in a letter dated November 23, 1998 (Ref. 2). The changes would allow the licensee to increase the allowable safety/relief valve (SRV) as-found setpoint tolerance from ±1% to ±3%.

The supplemental information contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original Federal Register notice.

2.0 BACKGROUND

10 CFR Part 50, Appendix A, General Design Criterion 15, "Reactor coolant system design" states that "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

The proposed change does not alter the SRV safety lift setpoints, relief setpoints, the SRV lift setpoint test frequency, or the number of SRVs required to be operable. Also, the proposed change requires the as-left safety valve function settings to be within ±1% of the specified nominal lift setpoints prior to installation before testing. The staff has previously granted approval to individual BWRs to increase the as-found SRV tolerance to three percent. The basis for the approval was a staff safety evaluation report (SER) for a licensing topical report (LTR NEDC-31753P) evaluating the setpoint tolerance increase. The staff SER (Ref. 3) included six conditions which must be addressed on a plant-specific basis for licensees applying for the increased SRV setpoint tolerance:

(a) Transient analysis of all abnormal operational occurrences as described in NEDC-31753P (Ref. 4), should be performed utilizing a ±3% tolerance for the safety mode of spring safety

valves (SSVs) and SRVs. In addition, the standard reload methodology (or other method approved by the staff) should be used for this analysis.

(b) Analysis of the design basis over-pressurization event using the 3% tolerance limit is required to confirm that the vessel pressure does not exceed the ASME pressure vessel code upset limit.

(c) The plant-specific analysis described in items (a) and (b) should assure that the number of SSVs, SRVs, and relief valves (RVs) included in the analyses correspond to the number of valves required to be operable in the technical specification.

(d) Reevaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc,), motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the 3% tolerance limit.

(e) Evaluation of the ±3% tolerance on any plant-specific operating modes (e.g., increased core flow, extended operating domain, etc.) should be completed.

(f) Evaluation of the effect of the 3% tolerance limit on the containment response during loss of coolant accidents and the hydrodynamic loads on the SRV discharge lines and containment should be completed.

3.0 EVALUATION

The safety objective of the SRVs is to prevent over-pressurization of the nuclear system. This protects the nuclear system process barrier from failure which could result in the uncontrolled release of fission products. The pressure relief system at Perry includes nineteen SRVs, arranged into three setpoint groupings: one group of SRVs (8) set at 1165 psig, a second group of SRVs (6) set at 1180 psig, and a third group of SRVs (5) set at 1190 psig. Existing TS provides a ±1% as-found tolerance and ±1% as-left setpoint tolerance. The proposed modifications would provide a ±3 % as-found tolerance and ±1% as-left setpoint tolerance. The licensee's submittal was evaluated against the generic SER described above.

3.1 Transient Analysis / Reload Methodology

The licensee must consider the impact of the tolerance increase on abnormal operational transients (AOTs). For Perry, analysis (cycle 7 reload analysis) of AOTs has been conducted utilizing the 3% tolerance and with 17 of the total 19 SRVs in service. All future reload analyses are expected to assume the 3% tolerance. The transient which generates the limiting decrease in a critical power ratio is the load rejection without turbine bypass event. The analysis showed that the thermal limits of the limiting transient would not be affected by the relaxation of SRV setpoint tolerance. Further, other transient events remain non-limiting and bounded by the above event. The staff finds the licensee's analysis acceptable because it was performed using a methodology previously approved by the NRC (Ref. 5).

3.2. Analysis of the Design Basis Overpressurization Event

The licensee is required to reevaluate the limiting design-basis pressurization transient using the 3% tolerance limit to confirm that the vessel pressure does not exceed the American Society of Mechanical Engineers (ASME) pressure vessel code upset limit. The ASME Boiler and Pressure Vessel Code Section III permits pressure transients up to 10% over design pressure (110% x 1250 psig = 1375 psig). The limiting pressurization AOT analyzed is a main steam isolation valve (MSIV) closure event occurring at the end of full power life without credit for a reactor trip on MSIV position sensing. The licensee analyzed (Ref.1) the MSIV closure event using the staff-approved model ODYN with the 3% tolerance and calculated the maximum vessel pressure to be 1289 psig. This is within the 1375 psig ASME limit, and is acceptable to the staff.

3.3. TS Operability Statement for SRVs

The licensee has stated that plant-specific overpressure analyses (Ref.1) have been conducted with the number of SRVs included in the analyses corresponding to the number of valves required to be operable in TS. The analysis took credit only for 13 of the 19 SRVs required by the TS. This is acceptable to the staff.

The surveillance frequency of the SRVs is specified in the plant TS to be in accordance with the plant in-service testing (IST) program. The IST program is required to meet the ASME Code which specifies that the SRVs must be tested at least every 5 years. However, the licensee stated that the current licensing basis for Perry is that at least half of the SRV population is removed and tested each refueling outage. This test frequency is sufficient to meet the test frequency specified in the staff SER (Ref. 3) for LTR NEDC-31753P, and is acceptable.

3.4. Reevaluation of the Performance of High Pressure Systems

The licensee must also reevaluate performance of high pressure systems (pump capacity, discharge pressure, etc.), considering the 3% tolerance limit. Perry has three systems which are required to inject to the vessel at high pressure conditions: high pressure core spray (HPCS), reactor core isolation cooling (RCIC), and standby liquid control system (SLCS). The most significant impact is the increased reactor pressure specified for system operation. The systems' performances were evaluated for the new reactor pressure of 1200 psig from 1177 psig. The HPCS system was determined to have the capability to inject its design flow of 517 gpm to the vessel at the new maximum pressure of 1200 psig without any changes. The RCIC turbine maximum steam flow rate is increased from 34,200 lbm/hr to 34,800 lbm/hr. The RCIC turbine/pump maximum speed is increased from 4550 rpm to 4600 rpm in order for the RCIC system to perform at the new maximum reactor operating pressure. The increased speed reduces the over-speed margin from 125% to 122.3%. This reduction in margin is acceptable due to the system modifications to the turbine start feature. The SLCS system was determined to have the capability to inject boron into the vessel at its design flow rate at the higher reactor pressures.

3.5. Evaluation of Motor-Operated Valves and Piping

In support of the SRV tolerance increase from ±1% to ±3%, the licensee stated that a maximum expected differential pressure calculation was performed for all valves in the Generic Letters (GL) 89-10 and 96-05 program for various operational conditions. The licensee determined that for dynamic and static testing of the MOVs, there is no effect on the maximum expected differential pressures resulting from the SRV safety setting tolerance increase. The staff finds that meeting the requirements of the GLs 89-10 and a 96-05 program is sufficient regarding required operational capability of the MOVs. The licensee further stated that there are no safety-related, air-operated, or hydraulically-operated valves whose functional capability would be adversely affected by the increased SRV tolerance, which is acceptable.

An increase in SRV setpoint tolerance involves a potential increase in SRV discharge hydrodynamic loads on the SRV discharge piping and quencher, the submerged structures, and the suppression pool boundary. The licensee reviewed the load increase to determine if sufficient conservatism and margins are available in the currently defined SRV loads. As a result, the licensee determined that the increase in SRV opening setpoint pressure would not adversely impact the current design-basis SRV hydrodynamic load analysis results.

The licensee also evaluated the effects of the high pressures associated with the increased setpoint tolerance on the instrumentation and piping for the systems. The licensee determined that no changes to instrumentation will be required. The licensee also determined that the impact of the higher pressure on system piping and other components was negligible.

The staff believes that the licensee has performed the appropriate analysis to determine any adverse impact of the proposed changes on motor operated valves and piping. The staff has reviewed the methodology used by the licensee for the above evaluations and results and concludes that it is acceptable.

3.6. Alternate Operating Modes

The licensee must also evaluate the increased tolerance on any plant-specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) The analyses included evaluations for the currently approved operating domains: Maximum Extended Operating Domain (MEOD), Increased Core Flow and Single Loop Operation. The analyses were found acceptable by the staff.

3.7. Containment Response / Hydrodynamic Loads

As previously described, the pressure relief system at Perry includes 19 SRVs, arranged into three setpoint groupings: one group of SRVs (8) set at 1165 psig, a second group of SRVs (6) set at 1180 psig, and a third group of SRVs (5) set at 1190 psig. The tolerance level of ±1% was small enough to have maintained three distinctive bands. The inference being that during transient conditions, the SRVs would be expected to open in discrete groupings, or bands, as opposed to all 19 SRVs opening at once. This effect can be seen by looking at the minimum and maximum values of acceptable lifting SRV pressures. The following array provides this information for each of the three distinctive setpoints:

1165 psig with an acceptable range of 1153.3 psig to 1176.6 psig
1180 psig with an acceptable range of 1168.2 psig to 1191.8 psig
1190 psig with an acceptable range of 1178.1 psig to 1201.9 psig

As seen above, the existing SRV setpoint tolerances result in minimal overlap between the three setpoint settings. While banding is generally retained for the setpoint tolerance of ±1%, increasing the setpoint tolerance to ±3% allows for more significant overlap of tolerances. A similar presentation including ±3% tolerances produces the following results:

1165 psig with an acceptable range of 1130.1 psig to 1199.9 psig
1180 psig with an acceptable range of 1144.6 psig to 1215.4 psig
1190 psig with an acceptable range of 1154.3 psig to 1225.7 psig

As depicted above, if the SRVs setpoint were allowed to drift such that their actual setpoints approached the ±3% tolerance limit, sufficient setpoint overlap could exist such that the concept of banding would be lost. As stated in the licensee's submittal, when using the proposed acceptance criterion of ±3%, an individual SRV would not need to be recalibrated provided the as-found setpoint was found to be within the ±3% tolerance limit. However, if the as-found setpoint was found to be outside the ±3% tolerance limit, the SRV would be recalibrated to within ±1%.

The original design for the Perry SRVs assumed a limited number of SRV actuations for any given sequence. Given unchecked setpoint drift, the current request could mathematically result in the actuation of all SRVs for a single sequence as opposed to a discrete number of SRVs. The concern is whether simultaneous actuation of most or all of the SRVs would violate any of the original licensing basis. Specifically, the staff questioned whether the limiting structural loading analysis assumed that all 19 SRVs opened simultaneously. In this regard, the licensee has confirmed that the limiting structural loading calculations have assumed that all 19 SRVs open at the same time and in phase. USAR Figure 5.2-6B shows the reactor vessel pressure transient for the case of all MSIVs closing at full power and indicating that all 19 SRVs open at once.

In summary, the licensee has analyzed the structural loading assuming the worst case scenario in which all 19 SRVs opened simultaneously and determined that the resultant hydrodynamic loadings were within acceptable limits. Therefore, the staff finds this acceptable.

3.8. ECCS-LOCA

GE reviewed the LOCA analysis in the Perry USAR for the licensee to determine the effect of an increase in SRV opening pressures on ECCS performance. The limiting break LOCA, the DBA recirculation break, the small break LOCA, and the steam line break outside containment events were evaluated to determine the effects of the increased SRV setpoint tolerance. For the six SRVs equipped with Low - Low Set (LLS) logic, the increased SRV safety mode setpoint to +3% assumed for postulated small break LOCAs will only affect the timing of the first actuation. Once the logic is initiated, the opening and closing setpoints of these preselected SRVs are automatically reset to lower values by the LLS logic. This logic is not affected by the setpoint tolerance change since it acts on the relief mode of operation and not on the safety mode of

operation. The acceptance criteria given in 10 CFR 50.46 are still satisfied for all break sizes and locations and hence the setpoint tolerance change for LOCA considerations is acceptable.

3.9. ATWS

The main steam isolation valve closure under ATWS conditions was reevaluated to support the tolerance increase of 3%. Using the staff-approved ODYN code and assuming two SRVs inoperable, the analysis shows that the vessel pressure reaches a maximum of 1344 psig, which is within the vessel overpressure criterion of 1500 psig for ATWS events. The long term effect on suppression pool temperature due to 3% SRV tolerance is negligible because there is little change in the total energy discharged to the pool. The staff finds this acceptable.

3.10 TECHNICAL SPECIFICATION CHANGES

The setpoint tolerance in TS 3.4.4 is changed from ±1% to ±3%. This is acceptable as described in this SER.

The following note is added to TS Bases page B 3.4-21: " The safety lift setpoints will still be set within a tolerance of + or - 1 % , but the setpoints will be tested to within + or - 3 % to determine acceptance or failure of the as found valve lift setpoint." This change is acceptable to the staff as described in this SER.

By letter dated July 13, 1998, FirstEnergy submitted proposed changes to the Perry Technical Specifications. The proposed amendment will allow the licensee to increase the allowable SRV setpoint tolerance from ±1% to ±3% . In support of the modifications, the licensee has submitted plant specific analyses adequately addressing the six conditions identified in the Staff's SER for NEDC-31753P, " BWROG In-Service Pressure Relief Technical Licensing Topical Report." The proposed changes are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (63 FR 43214). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: George Thomas
Gary Hammer
Amira Gill

Date: March 3, 1999

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Original signed by:
Douglas V. Pickett, Senior Project Manager
Project Directorate III-2
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Docket No. 50-440

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*See G. Thomas to D. Pickett memo dated 01/15/99

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